§34. Erosion/deposition and Hydrogen Retention Study on the Tungsten Divertor Tiles in LHD

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Tungsten is a potential candidate for the divertor armor materials due to excellent thermal properties, low sputtering yield and low retention properties for hydrogen isotopes. However, undesired tungsten accumulation has been observed on some tokamak machines [1-3]. In this study, therefore, four tungsten coated graphite divertor tiles by using vacuum plasma splay (VPS) method with the thickness of 100 μm were installed in the LHD (Large Helical Device) in 2008FY campaign. Four tiles were installed at the lower, outer and inner position of the helical divertor array. Tungsten emission from the tungsten surface and reliability of the coated tungsten layer and graphite (IG-430U, Toyo-tanso) substrate has been investigated by using the real-time spectroscopic analysis and the post-mortem surface analysis, respectively.

The typical edge plasma parameters of electron temperature, electron density and heat flux in LHD are T_e~20-30 eV, n_e~10^{18} m^{-3} and Γ~1-3 MW/m^2, respectively. For the real-time analysis of the sputtering erosion of the tungsten, spectroscopic measurement interested in WI (λ=400.9 nm) line had been performed with a visible spectrometer. The line of site of the spectrometer was adjusted on the inner VPS-W/IG tile, but the WI line could not clearly identify during the discharges. However, tungsten depositions with the amount of 5.6×10^{20} W/m^2 were identified on the graphite tile surface which was located just next to the VPS-W tile. This means that WI emission from the surface seems to be below the detection limit of the spectroscope but sputtered tungsten atoms were at least not zero.

After the campaign, all tiles were taken out and their surfaces were checked. Fig. 1 shows after exposed VPS-W tile using at the inner position of the helical divertor array. Although the strike points of the divertor plasma where heat and particle loads were highly localized suffered sputtering erosion, all VPS-W/IG tiles were sufficiently withstood and were observed no macroscopic damages for single experimental campaign (6509 plasma shot). The central part of this tile has still kept its original tungsten color, while, side parts were changed to black. From the energy dispersive X-ray spectroscopy (EDS) analysis on the side parts, mixed-material depositis composed by C, O, Fe and W was identified (majority of the deposited element is C). This indicates that erosion (central part) and deposition (side parts) locally occurred even in a single tile. It is the first information taken from the tungsten tile and such erosion/deposition pattern is consistent with the graphite tile case. To confirm the nano-scale surface modification caused by the divertor plasma, TEM observation was applied with focused ion beam (FIB) fabrication technique. Fig. 2-(a) -(b) shows cross-sectional TEM image of the erosion and deposition area. In the case of the erosion dominant surface, heavy damages such as large bubbles with size of about 1-20nm, dislocation loops and surface roughening occurred in the sub-surface region (~40 nm thickness). These intensive damages seem to be mainly caused by helium ion incidence because it is known that injected helium atoms aggregate by themselves and form helium-vacancy complexes and helium bubbles. In the 2008FY campaign, some shots with helium gas puffing and 197h helium GDC were performed. Judging from the size, density and depth distribution of the bubbles and dislocation loops, surface temperature seemed to be raised at least over 1000°C. Such bubbles can be seen only on the erosion-dominated surfaces, but not seen in the boundary between deposited layers and tungsten surfaces, which suggests that the deposited layers formed in the initial conditioning discharges prevents damages by plasma discharges afterword. In the case of the deposition-dominant surface, upper half of the Fig. 2-(b) corresponds to the C dominant deposition layer. From the elastic recoil detection analysis (ERD), retained hydrogen is lowest at the erosion dominant area (0.73~2.4×10^{20} H/m^2), which are almost same value with the saturation level of the bulk tungsten. While, in the deposition dominant area, where were covered by C dominant deposits, hydrogen retention is about ten times higher (3.6~3.8×10^{21} H/m^2) than that of the erosion dominant area. This result indicates that if we would want to suppress the hydrogen co-deposition with carbon, full tungsten divertor system would be effective.

Fig. 1. After exposed VPS-W tile using at the inner position of the helical divertor.

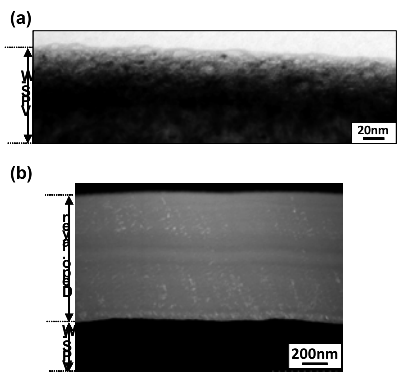

Fig. 2. Cross-sectional TEM image of the erosion dominant area (a), and deposition dominant area (b).

[1] Neu, R. et al., J. Nucl. Mater. 367-370 (2007) 1497
[2] Tanabe, T. et al., J Nucl. Mater. 283-287 (2000) 1128
[3] Nakano, T. et al., Nucl. Fusion 49 (2009) 115024